OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE RENEWAL OF

THE ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

TRIGA REACTOR FACILITY

LICENSE NO. R-84; DOCKET NO. 50-170

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of the Armed Forces Radiobiology Research Institute (AFRRI) license renewal application dated June 24, 2004 (a redacted version of the safety analysis report (SAR) is available on the NRC's public Web site at www.nrc.gov under Agencywide Documents Access and Management System (ADAMS) Accession No. ML041800067), as supplemented. The NRC staff's review used the guidance in NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," and supporting information from the American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors." During our review, questions have arisen, for which additional information and clarification is needed. This request for additional information (RAI) identifies the additional information needed to continue our review. Many of these RAIs below will refer to the Technical Specifications (TSs) provided by the AFRRI letter dated February 26, 2016 (a redacted version can be found in ADAMS Accession No. ML16060A210). We request that you provide responses to this RAI within 30 days from the date of the cover letter.

1. For the purpose of re-issuing the AFRRI license, provide the total number of grams of uranium-235 and plutonium needed for special nuclear material, of any enrichment, in the form of detectors, fission plates, foils, and solutions.

    a. AFRRI currently holds Uranium 235 and plutonium in the form of detectors, fission plates and foils. Much of the current inventory is aging and will need to be replaced. AFRRI will require no more than 100 grams in the foreseeable future. If AFRRI's requirements exceed this amount in the future, a license amendment will be requested as the need arises.

2. Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 states that the NRC requires current confirmatory documentation, as a precondition to issuing or renewing an operating license for a research or test reactor, that the licensee has entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel.

    In the renewal application, AFRRI has not provided evidence of compliance with Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982.

Enclosure

Provide a copy of the current valid DOE fuel contract suitable for the NRC staff to use to verify that DOE continues to retain title of the AFRRI fuel and is obligated to take AFRRI's spent fuel and/or high-level wastes. Include the terms and effective dates of the agreement.

a. AFRRI is in communication with DOE and a new letter will be issued by DOE.

3. Proposed TS 5.1, "Site and Facility Description," does not include a description of the licensed area including the location of the exclusion or restricted areas.

   ANSI/ANS 15.1-2007, Section 5.1, "Site and Facility Description," guidance states, in part, that "A general description of the site and of the facility including location and exclusion or restricted areas shall be present."

   Provide a description of the licensed area including the location of the exclusion or restricted areas and refer only to publicly available documents (e.g., not the physical security plan), or provide an explanation of why it is not necessary. In addition, provide a description of the operations boundary.

   The following will be added to TS 5.1

   f. The reactor is housed in building #42 of the AFRRI complex and the restricted areas are located within that structure.

   The restricted areas are described in the SAR for the AFRRI reactor facility section 1.3.1 including figures 2-2 through 2-4 which describe the location of the reactor in the AFRRI complex. Figures 3-1 through 3-4 are the floor plan layouts which identify the reactor areas.

4. The compound adjective "long term" is not defined in proposed TS 4.6, "Reactor Fuel Elements," Specification a., which states:

   > Fuel elements shall be inspected visually for damage or deterioration and measured for length and bend in accordance with the following:
   >
   > a. Before being placed in the core for the first time or following long term storage;

   Quantify the compound adjective "long term" in proposed TS 4.6, Specification a.

   Long term is defined to be fuel that has been taken out of service with no plans for use for more than one fuel measurement cycle. The definition will be added to TS.

5. The inability to remove a TRIGA fuel element from the core is an abnormal condition that warrants further evaluation. The criteria for dispositioning fuel that cannot be normally removed through the upper grid plate is not addressed in proposed TS 3.7, "Fuel Parameters," Specification d., as this TS does not explicitly pertain to fuel that cannot be removed through the top of the grid plate.

   Modify proposed TS 3.7 to include fuel that cannot be removed through the top of the grid plate.

Tech Spec 3.7 FUEL PARAMETERS states specifically that "The reactor shall not be operated with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and removed from the core if:

a. The transverse bend exceeds .0625 inches …
b. The length exceeds its original length by 0.100 inches
c. A cladding defect exists as indicated by the release of fission products; or
d. **VISUAL INSPECTION IDENTIFIES BULDGES, GROSS PITTING, OR CORROSION**
e. The burnup of uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

TS 4.6 Reactor Fuel Elements states that all fuel in the core shall be inspected visually for damage or deterioration and measured for length and bend in accordance with the following:

a. before being placed in the core for the first time or following long term storage;

and depending on the location in the core, all fuel shall be removed, measured and inspected every 2-4 years. Furthermore, if a single element is found to be damaged, all fuel in the core shall be measured. Fuel cannot be measured and/or inspected if it has a bulge large enough to lock it into the grid plate, and the reactor shall not be operated if the fuel in the core has not been measured and inspected, so the question is bounded by existing TS 3.7 and 4.6. No modification to TS is required.

6. Proposed TS 3.2.1, "Reactor Control System," Specification b., states, "The reactor shall not be operated unless the four control rod drives specified in Section 5.2.2.b. are operable or fully inserted."

TS 3.2.1 b. should read:
The reactor shall not be operated unless the four control rod drives specified in section 5.2.2.b. are operable or the control rod associated with an inoperable control rod drive is/are fully inserted. TS 3.2.1 b. will be modified as stated

A. Clarify the conditions under which the reactor can be operated and the meaning of the term "fully inserted."

B. Clarify whether you meant that "[the associated control rod is] fully inserted," or that the control *drives must be* operable or *fully inserted*. The associated control rod is fully inserted.

7. Proposed TS 4.2.1, "Reactor Control Systems," Specification c., states:

> On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient rod system shall be performed. Semiannually, not to exceed 7.5 months, the transient rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary.

A. Define the phrase "functional performance check" and include the minimum performance criteria that is required to demonstrate proper operation of the transient rod, as well as the means for evaluating acceptable performance.

B. Indicate whether functional means a channel check or channel test. If it does not mean either of these, then define functional.

C. Modify proposed TS 4.2.1 to include these clarifications, or justify why the changes are unnecessary.

TS 4.2.1 should read: On each day that pulse mode operation of the reactor is planned, the transient rod system is channel tested to verify that the system is operable. Semiannually, not to exceed 7.5 months, the transient rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary.

TS 4.2.1 will be modified as stated

8. The following questions pertain to the limiting condition for operation, proposed TS 3.3, "Coolant Systems," and the corresponding surveillance requirement, proposed TS 4.3.

Proposed TS 3.3, "Coolant Systems," states, in part, the following:

Specifications

a. The reactor shall not be operated above a thermal power of 5 kW when the core outlet temperature exceeds 60°C;
b. The reactor shall not be operated if the conductivity of the bulk water is greater than 5 micromhos/cm; and
c. Both audible and visual alarms shall be provided to alert the AFRRI security guards and other personnel to any drop in reactor pool water level greater than 6 inches.

Proposed TS 4.3, "Coolant Systems," states, in part, the following:

Specifications

a. The pool water temperature, as measured near the input to the water purification system, shall be measured daily, whenever operations are planned.
b. The conductivity of the bulk water shall be measured monthly, not to exceed 6 weeks.
c. The reactor coolant shall be measured for radioactivity at least quarterly, not to exceed 4 months.
d. The audible and visual reactor pool level alarms shall be tested quarterly, not to exceed 4 months.

A. The guidance in ANSI/ANS 15.1-2007, Section 3.3, "Coolant Systems," states, "Minimum operating equipment, or operating limits, or both, shall be specified for the following: . . . (5) fission product activity detection; . . ."

However, no TS operating limit has been proposed on coolant radioactivity. Additionally, proposed TS 4.3, Specification c., is a surveillance for radioactivity detection, but no actions are specified to be taken if radioactivity is found to be above the operational limit.

Describe how abnormal coolant radioactivity (indicative of abnormal fission product activity) would be detected by the operator. Provide a TS limit on fission product activity that provides a maximum upper limit of acceptable activity. This TS should correspond with surveillance TS 4.3, Specification c. Specify the corresponding actions to be taken if the limit is reached. Alternatively, justify why it is not necessary.

Abnormally high levels of activity in the water would be detected by the myriad of radiation detection equipment in the reactor room. i.e. RAMS, CAMS, Stack gas monitor etc. Since the water is never directly discharged to the environment it does not pose a risk to either reactor staff, the public or the environment. AFRRI Health Physics Department (HPD) personnel collect and analyze primary and secondary reactor water on a monthly basis. Any radioanalysis result or direct survey that exceeds an Action Level shall be brought to the attention of the RFD within 24 hours of the posting of results.

The following action levels are typical for pool water and secondary cooling water:

Reactor Pool Water

Type Analysis	AL 1 (pCi/ml)	AL 2 (pCi/ml)
Gross Alpha	0.1	1
Gross Beta	0.1	1
Activation Products	0.1	1
Fission Products	*	*

* The presence of any fission product shall be brought to the immediate attention of the RSO and RFD. Fission products include: Kr-85, Sr-89, Sr-90, Sr-91, Sr-92, Y-90, Y-91, Zr-95, Nb-95, Nb-95m, Nb-97, Mo-99, Ru-103, Ru-106, Rh-103, Rh-106, Ag-111, Sn-125, Sb-125, Sb-127, Te-127, Te-127m, Te-129, Te-129m, Te-132, I-131, I-132, I-133, I-134, I-135, Xe-131, Xe-133, Xe-135, Cs-136, Cs-137, Cs-138, Ba-137, Ba-140, La-140, Ce-141, Ce-143, Ce-144, Pr-143, Pr-144, Nd-147, Pm-147

Secondary Cooling Water

Type Analysis	AL 1 (pCi/ml)	AL 2 (pCi/ml)
Gross Alpha	0.1	0.5
Gross Beta	0.1	0.5
Gamma Emitters	*	*

* The presence of any gamma emitter other than naturally occurring radionuclides shall be brought to the immediate attention of the RFD.

No change to TS is warranted.

B. The basis for the proposed TS 3.3 pool temperature limit discusses the protection of the water purification resins. Given that the thermal hydraulic analysis was based on the assumption of a maximum pool temperature of 60 degrees C, and given that reactor operation at temperatures greater than 60 degrees C would be unanalyzed, provide a revision to proposed TS 3.3 that specifies 60 degrees C as the maximum temperature permitted. Provide an appropriate basis for TS 3.3 that considers the results of the thermal hydraulic analysis. (See also RAI 31.D.)

TS 3.3 Coolant Systems will be changed to read:

Objective

a. To ensure the effectiveness of the resins in the water purification system;
b. To prevent activated contaminants from becoming a radiological hazard; and
c. To protect the integrity of the reactor core

Specifications

a. The reactor shall not be operated above a thermal power of 5 kW when the bulk water temperature exceeds 55° C;
b. The reactor shall not be operated if the conductivity of the bulk pool water is greater than 5 micromhos/cm;
c. Both audible and visual alarms shall be provided to alert the AFRRI security guards and personnel to any drop in reactor pool water level greater than 6 inches.

Discussion: with the reactor cooling system secured, the pool temperature rise is 14.1° C per hour at a power level of 1 MW thermal. At 5 kW, the temperature rise is calculated to be .0705 degrees per hour of operation. At this power level the reactor

would need to operate at 5kW for more than 70 continuous hours of operation to reach the 60° C mark. Furthermore, analysis shows that full power operations are safe at 60°. At a power level of 1MW thermal, the fuel temperature reaches approximately 400 ° C, a temperature rise of approximately 380 ° C. At a power level of 5kW thermal, the fuel temperature remains at ambient water temperature. TRIGA fuel has been tested to be safe beyond 1000 ° C. The overarching objective is to protect the integrity of the fuel. If the reactor were operated at 5kW indefinitely with the cooling system secured the worst case would be a fuel temperature at the maximum ambient water temperature of 100 ° C. The published safety limit for the AFRRI TRIGA is 1000 ° C. This leaves a safety margin of 900 ° C, assuming that no repairs have been made to the cooling system during this postulated operation. These TS provides more than reasonable assurance that safe operations are maintained.

C. The current TS 4.3, Specification b., states that conductivity shall be measured weekly. In your February 9, 2016, RAI responses (ADAMS Accession No. ML16040A310), you provided a justification for monthly measurement of conductivity, which stated:

> The stability of conductivity within the AFRRI TRIGA pool water system has been proven by more than 5 decades of operations. Furthermore, experience demonstrates that the conductivity of the pool water not vary with reactor usage. Additionally, corrosion is an extremely slow process, making daily/weekly measurements unnecessary. NUREG-1537, Part 1, Appendix 14.1, Section 4.3, "Coolant Systems," Item (6) "Conductivity and pH," provides guidance that the conductivity and pH should be measured weekly. Monthly measurements are permitted if the reactor is shutdown for long periods of time and/or if justification is provided in the SAR. Since conductivity is not a function of usage, and NUREG-1537 permits monthly measurements, then it should be acceptable to make measurements on a monthly basis, whether or not operations are planned.

The regulations in 10 CFR 50.36 cover the protection of fuel cladding. Describe alternate indications available to operators for early detection of ion exchange failure or other inadvertent contamination in the pool water, given a monthly surveillance requirement.

The AFRRI TRIGA core is comprised of stainless steel clad fuel. Corrosion is generally not an issue with SS clad fuel. Furthermore, corrosion is not a flash process, so that monthly measurements will provide reasonable assurance that the resins are operating within normal parameters. Attachments 1-6 show conductivity vs time over the course of 5 years of normal operations. During the sample 5 year period of time, resins were changed out at intervals ranging from 36 to 99 weeks. During this time frame, the conductivity rarely exceeded 50% of the allowed value. Given the proven stability of the AFRRI system, sampling intervals of 90 days should be sufficient to protect the integrity of the system. The proposed sampling interval of monthly would appear to be excessive; however AFRRI will accept monthly

surveillance for conductivity measurements. No change to the proposed TS surveillance is warranted.

9. There are inconsistencies between the specifications included in proposed TS 3.2, "Reactor Control and Safety Systems," and the corresponding surveillance requirements in proposed TS 4.2, "Reactor Control and Safety Systems." As stated in ANSI/ANS 15.1-2007, Section 4, "a specific system from a Section 3 specification will establish the minimum performance level, and a companion Section 4 surveillance requirement will prescribe the frequency and scope of surveillance to demonstrate such performance." Provide information that demonstrates that each function, scram or interlock has a corresponding surveillance, and ensures that each of the following issues are addressed.

A. Proposed TS 4.2.2, Specification c., states, "Channel calibration shall be made of the power level monitoring channels annually, not to exceed 15 months." However, no definition for power level monitoring channels is provided.

A definition for power level monitoring channel will be added to the definitions section:

POWER LEVEL MONITORING CHANNEL
A power level monitoring channel is defined to be a channel that is intended to provide real time power level readings to the operator.

List the specific neutron measurement instrumentation channels (linear power channel, log power channel and power pulsing channel) that constitute the "power level monitoring channels" as proposed in TS 4.2.2, Specification c. Provide a revision to proposed TS 4.2.2, Specification c., to clarify the definition of power level monitoring channels, or state why it is not necessary.

TS 4.2.2 c will be modified to read Channel calibration shall be made of the NP, NPP, NM1000, (NLW, NMP – the manufacturer's direct replacement for the NM1000) or any other console instrumentation designated to provide direct power level information to the operator, annually not to exceed 15 months.

B. Proposed TS 4.2.2, Specification a., states, "A channel test of the scram function of the high-flux safety channels shall be made each day that the reactor is to be operated."

The measuring channels—linear power channel, log power channel and power pulsing channel—have a TS for calibration (proposed TS 4.2.2, Specification c.), but the linear power channel, log power channel and power pulsing channel do not have either a channel check or channel test for the protective functions which they provide.

Revise proposed TS 3.2 and 4.2 to demonstrate the checking and testing of the operability of the measuring channels or justify why no change to the TSs are needed. In addition, describe why a channel check for high flux safety channel is

more appropriate than a channel test.

The operational channel does NOT provide a protective function. There are no scrams associated with this channel. However, a k-excess measurement serves the same function as a channel test. A channel test of the operation channel (Linear and log channels) is done daily. TS 4.1.c. will be modified to read:

The core excess reactivity shall be measured each day of operation involving the movement of control rods, or prior to each continuous operation exceeding more than a day, and following any significant (>$0.25) core configuration changes. At a minimum excess reactivity shall be measured annually, not to exceed 15 months. This measurement is also a complete channel test of the linear power channel and log power channel.

C. Proposed TS 4.2.2, Specification b., states, "A channel test of each of the reactor safety system channels for the intended mode of operation shall be performed weekly, whenever operations are planned."

Define reactor safety system channels and identify which specific channels are included in this definition. State which specific reactor safety system channels are applicable in proposed TS 4.2.2, Specification b., or justify why no change to the TS is needed. In addition, explain what a channel check of the scram function is, considering the definition in proposed TS 1.4, "Channel Check."

TS 4.2.2 will be modified to replace "safety system channels" with "high flux safety channels" Channel check should read channel test. TS 4.2.2 will be modified to read channel test.

In addition, a definition 1.35 will be modified to read "A safety channel is a high flux safety channel with scram capability"

D. State whether proposed TS 4.2.2, Specification a., provides justification for the lack of inclusion of a channel check for fuel temperature safety channels, or revise the TS to include a channel check for fuel temperature safety channels.

Fuel temperature surveillance is covered in TS 4.2.3. No further modification or explanation is necessary.

10. The following questions pertain to proposed TS 3.2.2, "Reactor Safety System," and proposed TS 4.2.2, "Reactor Safety Systems."

A. Proposed TS 3.2.2, Table 2, "Minimum Reactor Safety Systems Scrams," requires a preset timer to initiate a scram of the reactor 15 seconds after the initiation of a pulse. Table 3, "Minimum Reactor Safety System Interlocks," requires an interlock to prevent pulsing when the reactor power level is 1 kW or above. This scram and

interlock system is described in Section 4.10, "Reactor Control Components," of the SAR.

The proposed TSs do not contain surveillance testing of the interlocks for the reactor safety system channels.

Provide information on the testing of the pulse timer scram and the pulse initiation interlock. Describe the surveillance testing of the functions and provide a surveillance TS, or justify why no TS is needed.

The 15 second timer is not a safety system and will be removed from table 2. The pulse initiation interlock listed in table 3 is tested each day pulse operations are planned.

A footnote to table 3 will be added stating "Reactor safety system interlocks shall be tested daily whenever operations involving these functions are planned"

B. In Section 4.11.3 of the SAR, "High Flux Safety Channels 1 and 2," mention is made of the high flux safety channels forming part of the scram logic circuitry. Included is a statement that during pulsing operation, when reactor power level, as measured by the high flux safety Channel 2, reaches the maximum pulse power level specified in TSs, a scram logic circuit is activated which caused an immediate reactor scram. Proposed TS 3.2.2, Table 2, "Minimum Reactor Safety System Scrams," does not provide a minimum performance level for the pulsing power scram. Explain its absence or make appropriate corrections to TS 3.2.2.

The pulsing operation on a TRIGA is controlled by the physics of TRIGA fuel. The initiation of a scram takes approximately ½ second from initiation to actuation. The duration of a pulse ranges from 10ms to 100ms depending on the amplitude of the pulse. Since the duration of a pulse is governed by the physics of the fuel, not by a latent scram, this scram has no purpose and is therefore not tested and no credit is claimed for this function.

11. Proposed TS 6.2.5, "Audit Function," Item f., states the following: "Reactor Facility ALARA [as low as reasonably achievable] Program. This program may be a section of the total AFRRI program." Describe the "total AFRRI program" or revise this specification to clarify the requirements of this action.

Section 6.2.5 does not represent a specification. It is simply a list of programmatic functions for the auditor to review and does not need to be described in detail in this section.

12. In the December 4, 2014, RAI responses, AFRRI performed an analysis to calculate the Ar-41 dose to workers in the reactor bay. This analysis was based on a measured Ar-41 production rate at 1 MWt of 0.5 microcuries per second (with reactor at mid-pool). Also, in the December 4, 2014, RAI responses, AFRRI performed an analysis to calculate Ar-41 effluent release to the public. That analysis provided Ar-41 production rates for reactor operation at varying core positions. The lowest Ar-41 production rate was

0.0024 millicuries per kilowatt-hour (with reactor at core position 500), which would be equivalent to a production rate at 1 MWt of 0.67 microcuries per second. In the September 21, 2012, responses to RAIs, AFRRI performed an analysis of the dose to members of the public from the radiation shine from an Ar-41 plume leaving the facility stack. That analysis assumed an Ar-41 production rate of 5.1 microcuries per second.

Clarify the differences in the assumptions made and/or conditions required for each of these three Ar-41 production rates. Additionally, explain why the production rate of 0.5 microcuries per second is conservative for the calculation of worker doses in the reactor bay, given that the licensed power of AFRRI is 1.1 MWt and that Ar-41 production increases when the reactor core is at a location other than mid-pool.

> Argon is produced as a result of reactor operations in 3 areas. Mid pool, ER1 and ER2. The maximum dose to reactor staff attributed to argon production is mid pool in the water. Although argon is produced at higher levels when operating in ER1 or ER2, these areas are not accessible to personnel and the argon produced is exhausted out the AFRRI stack. Although the licensed power for the AFRRI reactor is 1.1 MW(t), the highest demand power is 1.0MW(t). for this analysis, if we assume a steady state power of 1.1MW(T), the argon production rate mid pool could be as high as 0.74 microcuries per second. This production rate translates to a SS concentration in the reactor building of 4.6E-7 microcuries/ml, and is well within the DAC.
>
> The answer to the Sept 21, 2012 RAI "Theoretical Dose to Members of the Public from AR-41 Release", the production rate is modified to be 9.9E-6 Ci/Sec to align with the TS production limit of 313Ci/year. This production rate would equal an exposure rate or .045mR/hr to the maximally exposed member of the public from the plume. If a 1/20th occupancy factor to a person sitting under the plume and that the wind direction changes 50% of the time in the direction of the hypothetical person the dose would be 9.86 mRem/year. This dose is ultraconservative and still remains below the 10mRem constraint level.

13. The NRC staff issued RAI 3b., on September 13, 2010, related to the regulations in 10 CFR Section 20.1301(a)(1). This question discussed limits to the total effective dose equivalent to individual members of the public likely to receive the highest dose from licensed operation. The question, RAI 3b., was as follows:

> Where is this person located? If this dose is from immersion in the Ar-41 plume when it reaches ground level, confirm that a higher dose is not possible from radiation shine from the plume passing over a person closer to the facility than the point at which the plume reaches ground level or from a person exposed to direct radiation shine from the Ar-41 source before release from the AFRRI research reactor facility.

In the September 21, 2012, RAI responses, AFRRI performed an analysis to calculate the theoretical dose to a person, exposed to direct radiation shine from the Ar-41 source released through the stack, standing 10 feet from the AFRRI exterior wall.

Clarify whether the person was assumed to be downwind and directly under the plume in a position maximally exposed to the shine from the overhead plume.

The postulated recipient of the calculated dose was assumed to be down wind and directly under the plume in a position maximally exposed to the shine from the overhead plume.

14. Proposed TS 3.5.2, "Effluents: Argon-41 Discharge Limit," Specification b., states:

> If calculations, which shall be performed at least quarterly but not to exceed 20 MWh of operation, indicate that argon-41 release in excess of 313.5 curies to the unrestricted environment could be reached during the year as a result of reactor operations, reactor operations that generate and release measurable quantities of argon-41 shall cease for the remainder of the calendar year.

Proposed TS 4.5.2, "Effluents," Specification c., states:

> A gaseous effluent release report shall be generated quarterly or every 20 MWh of reactor operation (whichever is first) to ensure radioactive effluent will not exceed the annual limit.

A. Proposed TS 3.5.2, Specification b. might be excessively conservative. Consider proposing an alternative TS which would provide greater operational flexibility, for example, reducing the number of hours to be operated instead of ceasing operation, while ensuring that Part 20 limits will not be exceeded.

TS 3.5.2 b. will be modified to state

If calculations, which shall be performed at least quarterly but not to exceed 20 MWh of operation, indicate that argon-41 release in excess of 313.5 curies to the unrestricted environment could be reached during the year as a result of normal reactor operations, reactor operations shall be curtailed for the remainder of the year as needed to ensure adherence with the 10 mRem constraint.

B. Clarify whether the gaseous effluent release report in proposed TS 4.5.2 would be based on actual measurement of Ar-41 releases, or on calculations of estimated Ar-41 releases based on operating history.

The gaseous effluent release report in proposed TS 4.5.2 is currently calculated from data collected during actual operations and is based on data collected during extensive measurements.

15. The regulations in 10 CFR Section 20.2003, "Disposal by release into sanitary sewerage," paragraph (a)(1) state that a licensee may discharge licensed material into

sanitary sewerage if, among other conditions, the material is readily soluble (or is readily dispersible biological material) in water."

Section 3.4.2 of the SAR, "Liquid Radioactive Wastes," does not explicitly state compliance with 10 CFR 20.2003. Discuss AFRRI's compliance with 10 CFR Section 20.2003 (a)(1).

All waste is transferred to the AFRRI O2 license and is comingled with all liquid radioactive waste within the AFRRI complex. Radioactive waste is collected, monitored and disposed of IAW 10CFRpart 20.2003 (a)(1) under the O2 license such that all liquid effluent release is compliant with 10CFRpart 20.2003 (a)(1). No liquid waste is disposed of under the R-84 license.

16. Proposed TS 3.5.1, "Monitoring System," states, in part, the following:

Specifications

The reactor shall not be operated unless the following radiation monitoring systems are operable:

a. Radiation Area Monitoring System. The radiation area monitoring (RAM) system shall have two detectors located in the reactor room and one detector placed near each exposure room plug door to detect streaming radiation;
b. Stack Gas Monitor. The stack gas monitor (SGM) will sample and measure the gaseous effluent in the building exhaust system;
c. Continuous Air Particulate Monitor. The continuous air particulate monitor (CAM) shall sample the air above the reactor pool. This unit shall be sensitive to radioactive particulate matter. Alarm of this unit shall initiate closure of the ventilation system dampers, restricting air leakage from the reactor room; and
d. Table 4 specifies the alarm and readout system for the above monitors.

Table 4. Locations of Radiation Monitoring Systems

Sampling Location	Readout Location(s) (Audible and Visual)
RAM	
Reactor Room (2 require	Reactor and Control Rooms
Exp. Room 1 Area	Prep Area and Control Rooms
Exp. Room 2 Area	Prep Area and Control Rooms
SGM	
Reactor Exhaust	Reactor and Control Room
CAM	
Reactor Room	Reactor and Control Room

A. Describe how proposed TS 3.5.1 is inclusive of other activities, performed when the reactor is not operating (i.e., removing an experiment with reactor shutdown, handling of fuel, etc.), which have the potential for airborne radiological release.

TS 3.5.1 has been modified to read:

The reactor shall be secured unless the following radiation monitoring systems are operable:

a. Radiation Area Monitoring System:

   i. 2 RAMS on the reactor Deck (Room 3160) are operable

   ii. If operating in an exposure room (ER1 or ER2) the RAM adjacent to the exposure room in use shall be operable

b. Stack Gas Monitor: The stack gas monitor (SGM) shall sample and measure the gaseous effluent in the exhaust system;

c. Continuous Air Particulate Monitor: The continuous air particulate monitor (CAM) shall sample the air above the reactor pool. This unit shall be sensitive to radioactive particulate matter. Alarm of this unit shall initiate closure of the ventilation system dampers, restricting air leakage from the reactor room; and

d. Table 4 specifies the alarm and readout system for the above monitors.

TS 3.6 .a. states:

If the possibility exists that a release of radioactive gasses or aerosols may occur, the amount and type of material irradiated shall be limited to ensure yearly compliance with table 2, Appendix B, of 10CFR Part 20, assuming that 100% of the gasses or aerosols escape. This question is bounded by TS 3.6. and definitions 1.32 and 1.33. No further restrictions or changes to TS 3.5.1 are warranted.

B. The guidance in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, "Technical Specifications," Section 3.7, "Radiation Monitoring Systems and Effluents," contains Table 14.4, "Typical required radiation measuring channels," with illustrative setpoint value maximums.

Proposed TS 3.5.1 does not include applicable alarm or actuation setpoints for the RAM, CAM, and SGM. Provide a revision to TS 3.5.1 that includes applicable alarm or actuation setpoints for the RAM, CAM, and SGM and an appropriate basis.

Radiation alarm setpoints are not typically included in the technical specifications (See Reed Research Reactor and Texas Engineering Experiment Station technical specifications as examples of recently issued licenses). Setpoints are set to ensure ALARA for radiation workers. ALARA takes into account background levels during operations, internal exposure limits set by the radiation safety group and best practice, At AFRRI, reactor bay RAMS are typically set to 10 mrem, CAMs to 40000 CPM and the SGM setpoint is calculated annually based on part 20 off site exposure rates. Reactor staff members are badged, and radiation exposures are controlled as required by 10CFR part 20. No further changes are warranted to TS 3.5.1.

CAM locations and Settings

Location	Alarm Level (CPM)	Alert Level (CPM)	Chart Recorder
Reactor Deck-Primary	40K	20K	On
Reactor Deck-Backup	40K	20K	Off
Prep Area	2K	1K	Off
Exposure Room 1	50K	25K	Off
Exposure Room 2	50K	25K	Off
ERT- Primary	2K	1K	Off

ERT- Backup	2K	1K	Off

During emergency conditions the RSO/RFD may direct the recorders on additional CAMs be activated.

17. The following questions pertain to proposed TS 6.6, "Operating Reports":

A. Proposed TS 6.6, "Operating Reports," Item b.7.a., requires liquid waste discharges to be summarized on a quarterly basis.

The regulations in 10 CFR 20.2003(a)(2) state that a licensee may discharge licensed material into sanitary sewerage if, among other conditions, the quantity of licensed or other radioactive material that the licensee releases into the sewer in one month, divided by the average monthly volume of water released into the sewer by the licensee, does not exceed the concentrations listed in Table 3 of Appendix B to 10 CFR Part 20.

Revise TS 6.6, Item b.7.a., to require monthly averaging of liquid waste discharges to ensure compliance with 10 CFR 20.2003(a)(2), or justify why no revision is needed.

TS 6.6 item b.7.a will be changed to summarized on a monthly basis

B. Proposed TS 6.6, "Operating Reports," Item b.7.a.i., states the annual report will contain, for liquid waste, "Concentration limits used and isotopic composition if greater than 3×10^{-6} microcuries/ml for fission and activation products."

Provide an explanation of the meaning of this requirement, and an explanation and justification for the value 3×10^{-6} microcuries/ml for fission and activation products.

AFRRI utilizes a warm and hot waste collection system that comingles all liquid radioactive effluents. The collective waste from the AFRRI complex is measured and released in compliance with 10CFR20.2003(a)(2) and WSSC and EPA requirements. This is handled entirely under the O2 license. No liquid wasted is analyzed, processed or disposed of under the R-84 license. 6.6 b.7.a.i will be modified to read:

i. Radioactivity discharged during the reporting period:

Total radioactivity released (in curies);

Concentration limits used and isotopic composition for fission and activation products.

Total radioactivity of each nuclide released during the reporting period and, based on representative isotopic analysis, average concentration at point of release during the reporting period;

C. Proposed TS 6.6, "Operating Reports," states, in part:

> ...the following reports shall be submitted to USNRC Office of Nuclear Reactor Regulation unless otherwise noted: . . . b. Annual Operating Report: . . . Each annual report shall include: . . . 7. A summary of the nature and amount of radioactive effluents released . . .
>
> c. Solid Waste (summarized on a quarterly basis)
>
> Total cubic feet of atomic number 3 to 83 materials in solid form disposed of under license R-84;"

Explain and provide a regulatory basis for why proposed TS 6.6 Item b.7., Specification c., applies only to atomic number 3 to 83 materials, or expand the applicability of this proposed TS if appropriate. In addition, expand the proposed TS to include the total activity (in Curies), as well as the total volume, disposed under license R-84.

TS 6.6.7.c will be changed to read:

Total cubic feet and combined activity in curies of materials in solid form disposed of under license R-84.

18. In the January 17, 2012, responses to RAIs, AFRRI analyzed doses to AFRRI facility occupants ("Receptors A, B, and C") and workers in the reactor room for the maximum hypothetical accident involving failure of a fueled experiment. AFRRI's analyses assumed that a portion of the halogens released from the fueled experiment would plate out within the reactor room, and would not be available for release to the environment. It is not clear whether the external dose contribution to workers in the reactor room, and to AFRRI facility occupants, from halogens that are plated out within the reactor room was considered in these analyses.

Clarify whether the doses provided in Tables 4, 5, and 6 of the RAI responses dated January 17, 2012, included the external dose contribution from halogen plate out.

Yes.

19. NUREG-1537, Part 1, Chapter 13, Section 13.1.3, "Loss of Coolant," provides guidance to licensees to systematically analyze and discuss credible accidents in each accident category.

AFRRI's revised SAR Chapter 13 (provided to NRC by letter dated March 4, 2010), Section 13.2.1.4, describes the radiation dose rates in the reactor floor and roof areas

due to the unshielded reactor core after a postulated large loss-of-coolant accident (LOCA).

In its RAI dated July 19, 2010, the NRC staff requested that AFRRI provide accumulated doses to the reactor building occupants and to the maximally exposed member of the public, considering evacuation procedures and potential residence time for staff, and asked AFRRI to discuss compliance of these doses with the regulations in 10 CFR Part 20.

AFRRI's response, dated February 7, 2011, provided accumulated doses, estimated based on the analysis in an older version of the AFRRI SAR that was provided to NRC as part of AFRRI's initial license renewal application submittal in 2004.

Provide accumulated doses for the LOCA accident that are based on the analysis provided in AFRRI's revised SAR, Section 13.2.1.4, that was submitted to NRC on March 4, 2010.

The dose to a reactor staff member standing on the core carriage (very conservative) if the 15,000 gallons of pool water flashed to steam or vanished within 1 second post shutdown: Discussion: The AFRRI pool is aluminum lined surrounded by many feet of concrete, except for 2 small projections into an exposure room without a concrete barrier. In order to flash one pound of water at STP, 1112 BTUs of energy, (assuming 100% energy transfer) would be required. This equates to 133,000,000 BTUs (equivalent to the energy that would be expended from a blast from 30 tons of TNT) of energy expended within a one second period of time to remove the water and produce a dose rate of 126 Rem per hour to a now dead person (from the explosion) standing above the pool, approximately 5 meters from the core. The only way to produce this quantity of energy in this short of a time frame would be an act of sabotage or terrorism. This is beyond the scope of an accident analysis. The only credible LOCA for the AFRRI TRIGA would involve a large crack or hole in the core projection (nodes), which could potentially empty the pool within a short time frame. The nodes are protected by 12 feet of concrete shield doors.

After 6 inches of pool water drains from the pool, a building evacuation alarm is automatically sounded. Experience demonstrates that the reactor area can be evacuated within 1 minute and the building within 10 minutes. While operating, the reactor roof area is not accessible to personnel or members of the public. As the pool continued to drain (assuming that no action would be taken to mitigate the loss of water),the building would be evacuated before enough shielding would drain to produce measurable dose to the public within the AFRRI building. Any non AFRRI members of the public would be more than 60 meters from the reactor room, with several intervening shielding walls such that any dose to the public would be below part 20 limits (extrapolating from unshielded dose rate calculations from the 2010 SAR LOCA analysis at 20 and 30 meters. The Dose rate at 60 meters would be less than 0.065 mR/hr. In addition, there are 3 independent systems available to fill the pool in an emergency; the largest capacity system consists of a 4 inch fire hose at 100PSI which can be connected directly to the pool.

Assuming that the core is completely uncovered within 15 minutes post shutdown, the person standing on the core shroud would be standing in a radiation field approximately 28 R per hour. Assuming evacuation time from the 50 foot square room of 2 minutes, the dose received would be approximately 1 Rem. If, for the sake of conservatism one was to assume that a person was still able to respond after the event that flashed 15,000 gallons of water was to evacuate, the dose rate above the core was calculated to be 126 Rem per hour 1 second after shutdown. Assuming a 2 minute stay time above the reactor core, a dose of 4.2 REM would be received, still within the annual part 20 exposure limit for a radiation worker.

20. ANSI/ANS 15.1-2007 guidance contains a definition for confinement. The proposed TSs do not contain this definition. Provide a definition for confinement or justify why it is not necessary.

The ANSI definition for Confinement will be added to the definitions section or TS.

> Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

21. ANSI/ANS 15.1-2007 guidance contains a definition for control rods that includes the functions that control rods provide. The proposed TSs do not cover the functions of control rods. Provide a definition for control rods that includes the types of functions that control rods provide, or justify why it is not necessary.

The following definition will be added to the definitions section:

A control rod is a device fabricated from neutron absorbing material or fuel, or both, that is used to establish neutron flux changes and to compensate for routine reactivity losses. Scrammable control rods can be quickly uncoupled from their drive units to rapidly shutdown the reactor if needed.

22. Proposed TS 1.8, "Excess Reactivity," states:

> Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions or at a specific set of conditions.

Define what is meant by "or at a specific set of conditions" and why it is necessary to include in the TS.

TS 3.1.3 states that an excess measurement is performed at the beginning of each day that operations are performed .... Depending on the core history, significant xenon (more than $0.01 specified in the definition of Reference Core Condition) buildup could be present in the core. If reference core conditions were required a valid excess measurement would not be possible. No change to this TS is warranted.

23. Proposed TS 1.22, "Reactor Safety System," states:

> Reactor safety systems are those systems, including their associated input channels, that are designed to initiate a reactor scram for the primary purpose of protecting the reactor or to provide information for initiation of manual protective action.

Proposed TS 1.22 accords, in part, with ANSI/ANS-15.1-2007. However, ANSI/ANS 15.1-2007 also contains a definition of "protective action." The term "protective action" is not defined in the proposed TSs. Provide a definition for the term "protective action."

A definition of Protective Action will be add to TS.

Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified set point.

24. The AFRRI updated SAR, Chapter 7, dated September 27, 2010 (ADAMS Accession No. ML110260024), provides general information about the design of instrumentation and control systems, but does not describe specific details associated with the operation or reactivity control aspects of the servo system.

NUREG-1537, Part 1, Chapter 7.3, "Reactor Control System," provides guidance that the license should analyze the operation and performance of the system, including the bases for any technical specifications and surveillance requirements, and provide a description of the evaluation of any accident scenarios that may be created by a malfunction of the system (e.g., a malfunction of the servo bounded by another reactivity insertion event).

A. Provide details of the servo system operation including the normal reactivity control range, regulating rod position, interlocks, and any other significant design and safety information, or explain why no additional information is necessary.

The TRIGA servo control system is described in detail in the SAR chapter 7.3.2 "Servo System". All interlocks and scrams applicable to the manual mode of operation apply to automatic mode as well. Interlocks are described in section 7.3.3 of the SAR. No additional information is needed to describe these systems..

B. Explain if additional technical specifications are needed for the servo system, or justify why no changes are necessary.

The description of the servo system is complete as written. No additional TS changes are warranted.

C. According to proposed TS 1.33, "Steady State Mode," operation of the reactor can be manual or automatic. Provide a description of the number of rods that can be simultaneously withdrawn and the interlocks which prevent insertions of excess reactivity. If three control rods can be withdrawn simultaneously, analyze this rod withdrawal accident and discuss how the analysis affects the results of the analysis of a ramp insertion accident.

The definition of Steady State Mode is contained in the definitions section, 1.33. Simultaneous manual withdrawal of more than one control rod is not allowed under TS 3.2.2. Automatic mode allows the simultaneous servo of more than one control rod. Analysis of this mode of operation was provided April, 2012, Feb 2016. The following information is offered in addition to the information submitted April 20, 2012, Feb, 2016.

This analysis is related to a rod withdrawal accident in which three standard control rods: shim, regulating, and safety, can be withdrawn simultaneously. There are 2 redundant, independent power and scram channels. If the first scram channel failed, the second channel would terminate the reactivity insertion at the same power level, 1.09 MW(t). For this analysis, the 3 second period RWP is also ignored. There is a delay from the initiation of a scram to the insertion of a control rod of no more than 0.5 seconds. This is the time necessary to close relay contacts, and bleed the magnetic field from the rod drive magnetic coupling of a standard control rod. The average insertion rate of the three standard control rods is 0.2285 ($/sec). Starting at 1.0 MW(t), the reactor would initiate a scram at 1.09 MW, and the three standard control rods are assumed to continue driving out for 0.5 seconds, resulting in an additional reactivity insertion of $0.11, which would produce a positive period of 75 seconds.

At scram + 0.5 seconds when the insertion would be terminated and the rods all scrammed, the peak power would be 1,097,290.94 watts. The maximum temperature would be 416.98 °C assuming that peak power and temperature are reached instantaneously before scramming the reactor.

In Table 1, the second column represents the total rod worth of each standard control rod and the sum of the total rod worth of all three standard control rods combined. The third column represents the withdrawal time of each standard control rod and the average withdrawal time of three standard control rods. The fourth column represents the rod worth insertion rate of the standard control rods which is the quotient of rod worth divided by withdrawal time.

Standard Control Rod	Rod Worth ($)	Withdrawal Time (sec)	Insertion Rate ($/sec)
Safety	2.65	39.4	0.0673
Shim	2.74	36.1	0.0759
Regulating	3.01	34.8	0.0865
Safe/Shim/Reg	8.40	36.8	0.2285

Table 1. Rod worth, withdraw time and insertion rate of the standard control rods.

In Table 2, the second column represents the reactivity insertion 0.5 seconds after the reactor scrams from each standard control rod and the three standard control rods being withdrawn simultaneously. This is the product of the insertion rate from Table 1 multiplied by 0.5 seconds. The third column represents the period resulting from the reactivity insertion in first column based on the In-Hour equation. The fourth column represents reactor power in watts 0.5 seconds following a scram starting at 1.09 MW, based on the period in the third column. The fifth column represents the fuel temperature in degrees Celsius corresponding to the reactor power in the fourth column, which was calculated based on empirical data fitted to a fifth degree polynomial.

Standard Control Rod	$ (0.5 sec)	Period (sec)	P (0.5 sec) Watts	T (0.5 sec) °C
Safety	0.0336	320	1,091,704.46	415.48
Shim	0.0380	280	1,091,948.17	415.53
Regulating	0.0432	230	1,092,372.14	415.60
Safe/Shim/Reg	0.1142	75	1,097,290.94	416.98

Table 2. Rod worth, period, power, and temperature of the standard control rods.

This analysis results in a negligible effect the on the results of the analysis of a ramp insertion accident. As a result of this analysis, the maximum power reached by withdrawing all three standard control rods simultaneously for 0.5 seconds following a scram that was initiated at 1.09 MW would result on a reactor power level of 1,097,290.94 watts or 1.097 MW which remains 2,709.06 watts below the AFRRI TRIGA licensed power limit of 1.1 MW. The fuel temperature calculated at this power level corresponds to 416.98 °C. This temperature is 583.02 °C below the safety limit of 1000 °C and 183.02 °C below limiting safety system setting of 600 °C.

25. For proposed TS 3.2.1, "Reactor Control System," explain how the notes to Table 1 apply to the Pulsing Power Channel, or modify the notes to exclude applicability to the Power Pulsing Channel.

For all modes of operation the final stage of repair/calibration is a test operation. For the purposes of testing and calibration limited operations must be an allowed condition of operations in order to complete repairs and/or calibrations. For pulse mode operations 3 measuring channels with scram capability are normally available. Under test 2 channels are still operational providing ample redundancy.

26. For proposed TS 3.2.1, Reactor Control System," explain how the notes to Table 2 apply, or modify the notes.

Notes 1 and 2 from Table 2 have been deleted.

27. Proposed TS 3.2.2, "Reactor Safety System," Table 2, refers to the high flux safety channel; however, this term is not defined. Provide a definition for high flux safety channel.

A definition for High Flux Safety Channel will be added to TS

A high flux safety channel is a power measuring safety channel in the reactor safety system.

28. Proposed TS 3.2.2, "Reactor Safety System," Table 2, refers to the "emergency stop." However, "emergency stop" is not defined. Provide a definition of "emergency stop" and describe how an "emergency stop" differs from a manual scram.

A definition for Emergency Stop will be added to TS

Emergency Stop is a safety interlock designed to prevent or cease reactor operations. Emergency stop buttons are provided in Exposure Room 1, Exposure Room 2 and on the console.

29. Proposed TS 3.2.2, "Reactor Safety System," Table 3, lists the minimum reactor safety system interlocks, but does not discuss the prevention of an inadvertent pulse. Describe how application of air to the transient rod (an inadvertent pulse) is prevented from occurring.

For all control rods to include the pulsing rod, in steady state mode, the control rod drives must be driven fully down in order to initiate the withdrawal of the associated control rod. For the transient rod drive specifically, in the steady state mode of operation air cannot be applied unless the control rod drive is fully down. Table 3 will be modified to include a test of Air Applied to Trans Rod Drive while not down.

30. Proposed TS 3.2.2, "Reactor Safety System," Table 3, uses the terminology "operational channel," but "operational channel" is not defined. Define operational channel or state why it is not necessary.

The following definition will be added to TS

Operational Channel: The Operational Channel is a power measuring channel used during steady state and square wave operations.

31. Proposed TS 3.2.2, "Reactor Safety System," does not contain a scram for pool water temperature. Proposed TS 3.3, "Coolant System," has as the basis that the pool temperature limit is designed to protect the demineralizer beds in the water purification system.

    A. Explain how proposed TS 3.3, "Coolant System," prevents operation if the pool temperature is above 60 degrees C.

    B. Explain how an automatic or manual scram is initiated for pool temperature, or explain how this TS for pool temperature keeps the reactor within analyzed conditions.

    C. Explain if the pool temperature protects the core during operations, and is in accord with the assumptions used for the thermal hydraulics analysis.

    D. Proposed TS 3.3, Specification a., states that the reactor should not be operated above 5 kWt when the coolant temperature measured at the core outlet is greater than 60 degrees C. Explain, analytically, at what coolant temperature above 60 degrees C does 5 kWt operation reach a thermal-hydraulic limit and add that temperature limit to the TS; or remove the ability to operate above 60 degrees C from the TS.

    See the answer to question 8 B. above.

32. Proposed TS 3.4, "Ventilation System," states, in part:

> The reactor shall not be operated unless the facility ventilation system is operating, except for periods of time not to exceed two continuous hours to permit repair, maintenance, or testing. In the event of a release of airborne radioactivity in the reactor room above routine reactor operation and normal background values, the ventilation system to the reactor room shall be automatically secured via closure dampers by a signal from the reactor deck continuous air particulate monitor.

Proposed TS 4.4, "Ventilation System," states, in part:

> The operating mechanism of the ventilation system dampers in the reactor room shall be verified to be operable and visually inspected monthly, not to exceed 6 weeks.

A. Propose a TS requirement to maintain a controlled air pathway (negative pressure) of the reactor room with respect to the adjacent rooms and surrounding building when the reactor is operating and during a postulated accident, or explain why it is not necessary. If a TS requirement to maintain negative pressure in the reactor room by a controlled air pathway is necessary, then propose a surveillance requirement to confirm that negative pressure is present.

TS 3.4 will be modified to read:

a.The reactor shall not be operated unless the facility ventilation system is operating, except for periods of time not to exceed two continuous hours to permit repair, maintenance, or testing. In the event of a release of airborne radioactivity in the reactor room above routine reactor operation and normal background values, the ventilation system to the reactor room shall be automatically secured via closure dampers by a signal from the reactor deck continuous air particulate monitor.

b.The reactor shall not be operated in exposure room 1 or 2

1. if the relative air pressure in the exposure room in use is greater than the reactor prep area (room 1105) except for periods of time not to exceed two continuous hours to permit repair, maintenance, or testing when the dampers shall be closed.

or;

2. the prep area RAMS E5 and E6 are alarming.

The following will be added to TS 4.4.:

The relative air pressure in the exposure room to be used shall be verified to be negative each day operations in the affected exposure room are planned.

The reactor exhaust damper flow failure closure system shall be tested each day that reactor operations are planned.

B. Explain how continuous air monitor setpoints are determined and verified for operability. Explain if the operating mechanism of the ventilation system dampers is

verified to be operable with a valid signal from the radiation monitors to the dampers, causing the dampers to close.

The set points are determined by the Radiation Safety Officer. Damper closure is verified by an alarm signal introduced to the CAM by a check source on each day operations are planned.

C. Proposed TS 3.4, "Ventilation System," discusses operating conditions for which the facility ventilation system must be operational. Revise the TS to require the ventilation system to be operating when radiation material is being handled in the reactor room with the potential for airborne radioactive material or justify why the TS is not needed.

See 32A above

33. Proposed TS 4.0, "Surveillance Requirements," states:

> No surveillance requirements shall be deferred during normal reactor operational periods. Any surveillance requirements that cannot be performed due to a reactor outage shall be performed prior to resuming normal reactor operations.

ANSI/ANS 15.1-2007, Section 4, "Surveillance Requirements," states, in part:

> For each surveillance requirement (SR), it should be specified if the surveillance activity can or cannot be deferred during reactor shutdown. It should also be specified for those that can be deferred, which must be performed prior to reactor operations.

Revise proposed TS 4.0 to include for each surveillance requirement, whether it is possible to defer during shutdowns, and if it is required prior to resuming operations.

TS 4.0 will be changed to read:

Surveillance requirements may be deferred during reactor shutdown (except TS 4.4, TS 4.5.1 and TS 4.5.2) however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Deferred surveillance items shall be performed as soon as practical after reactor startup. Scheduled surveillance that cannot be performed with the reactor operating may be deferred until to a non-power time.

	TS	Possible to defer during shutdowns?	Required prior to operations?
1.	4.1 Reactor core parameters	Yes	Yes
2.	4.2.1 Reactor Control Systems	Yes	Yes
3.	4.2.2 Reactor Safety Systems	Yes	Yes
4.	4.2.3 Fuel Temperature	Yes	Yes
5.	4.2.4 Facility Interlock System	Yes	Yes
6.	4.3 Coolant Systems	Yes	Yes
7.	4.4 Ventilation Systems	No	N/A

8.	4.5.1 Monitoring System	No	Yes
9.	4.5.2 Effluents	No	N/A
10.	4.6 Reactor Fuel Elements	Yes	Yes

34. Proposed TS 5.3, "Special Nuclear Material Storage," states, in part:

All fuel elements not in the reactor core shall be stored and handled in accordance with applicable regulations. Irradiated fuel elements and fueled devices shall be stored in an array that *will* [emphasis added] permit sufficient natural convective cooling by water or air, and the fuel element or fueled device temperature *will* [emphasis added] not exceed design values. Storage shall be such that groups of stored fuel elements *will* [emphasis added] remain subcritical under all conditions of moderation and reflection in a configuration where k_{eff} is no greater than 0.90.

A. Provide TS 5.3 with "shall" statements, or justify why it is not necessary to do so.

Shall does not make grammatical sense in this context. No change to TS is warranted

B. Define the meaning of "groups" or delete the term from the proposed TSs.

The word groups will be deleted from TS.

35. Proposed TS. 6.1.1, "Structure," states, in part:

The organization of personnel for the management and operation of the AFRRI reactor facility is shown in Figure 1. Organizational changes may occur based on AFRRI requirements and will be depicted in internal documents.

The organization chart is part of the facility operating license, and therefore changes to the organization chart require a license amendment. Provide a TS consistent with 10 CFR 50.36c.(5), or explain why it is unnecessary.

TS 6.1.1 will be changed to read:

Organizational changes to the general AFRRI organizational structure may occur based on AFRRI requirements and will be depicted in internal documents, however, organizational changes to the management and operations of the reactor facility organizational structure shall not be made without a change to TS 6.1.1.

36. Guidance in ANSI/ANS 15.1-2007, Section 6.1.2, "Responsibility," states, in part:

Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 1. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adherence to all requirements of the operating license or charter and technical specifications.

A. Proposed TS. 6.1.2, "Responsibility," states, in part, that "The AFRRI Licensee shall have license responsibility for the reactor facility." The "AFRRI Licensee" has not been defined. Define who, by title, the licensee is, or explain why it is not necessary to do so.

B. Proposed TS. 6.1.2, "Responsibility," does not explicitly state who is responsible for meeting the conditions of the license, and who is responsible for the protection of public health and safety. Provide a revised proposed TS 6.1.2, that clearly states the individuals who are responsible for meeting the conditions of the license and who are responsible for the protection of public health and safety.

6.1.3 STAFFING will be modified to include the following paragraph in 6.1.3.1 a:

a. AFRRI Licensee

The AFRRI Licensee is the AFRRI Director. The Director of the AFRRI TRIGA has management responsibility for adhering to the terms and conditions of the AFRRI reactor license R-84, the AFRRI O2 byproduct license, the AFRRI Technical Specifications and for protecting the health and safety of the facility staff and public.

37. Proposed TS do not contain a section similar to ANSI/ANS 15.1-2007, Section 6.1.1, "Structure," which contains a description of the organization for the management and operation of the reactor facility, including Levels 1-4 and their responsibilities.

Provide a TS with a description of the responsibilities of Levels 1-4, or explain why no changes are necessary.

TS 6.1.1.1 will be modified to include the following:

Management Levels:

Level 1: AFRRI Director: Responsible for the facility license.

Level 2: Reactor Facility Director: Responsible for reactor facility operations and shall report to level1.

Level 3: Reactor Operations Supervisor: Responsible for the day-to-day operation of the reactor and shall report to Level 2.

Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3 for matters involving reactor operations.

38. Proposed TS 6.1.3.2, "Operations," Specification b., states, in part:

> A Senior Reactor Operator shall be present at the reactor during the following operations: . . . 2. Initial reactor startup and approach to power

Provide a definition for the term "initial startup" or state why it is not necessary.

The definition of initial startup and approach to power will be added to TS definitions.

INITIAL STARTUP

The first reactor startup and approach to power following fuel element relocation within the core and/or any significant (>$0.25) core configuration changes.

39. Proposed TS 6.1.3.3, "Training of Personnel," states:

> Training and retraining program shall be maintained, to ensure adequate levels of proficiency in persons involved in the reactor and reactor operations.

However, the TSs do not state that the training program shall be consistent with the requirements in 10 CFR 55.59 or the standards related to training in ANSI/ANS-15.4-2007. Propose a revision to proposed TS 6.1.3.3 to require that the training program shall be consistent with the requirements in 10 CFR 55.59 and/or the standards in ANSI/ANS 15.4-2007, or state why it is not necessary to do so.

TS 6.1.3.3 will be modified to read:

Training and retraining program shall be maintained to ensure adequate levels of proficiency in persons involved in the reactor and reactor operations. The training and retraining program will be consistent with the R-84 requalification plan.

40. Proposed TS 6.2.1.1, "Composition," does not specify the minimum number of members of the review and audit group (The Reactor and Radiation Facilities Safety Subcommittee (RRFSS)). The guidance in ANSI/ANS 15.1-2007, Section 6.2.1, "Composition and Qualifications," states, in part, that "The review and audit functions shall be composed of a minimum of three members. . . ." Propose a modification to

proposed TS 6.2.1.1 that specifies the minimum number of members, or justify why it is unnecessary.

TS 6.2.1.1 states:

Composition

a. Regular RRFSS Members (Permanent Members)

1. The following shall be members of the RRFSS:

a. AFRRI Radiation Safety Officer (RSO)

b. AFRRI Reactor Facility Director (RFD)

2. The following shall be appointed to the RRFSS by the AFRRI Licensee:

a. Chairman

b. One to three non-AFRRI members who are knowledgeable in fields related to reactor safety. At least one shall be a Reactor Operations Specialist or a Health Physics Specialist.

b. Special RRFSS Members (Temporary Members)

1. Other knowledgeable persons to serve as alternates in section 6.2.1.1.a.2.b above as appointed by the AFRRI Licensee.

2. Voting ad hoc members, appointed by the AFRRI Licensee to assist in review of a particular problem.

c. Nonvoting members as appointed by the AFRRI Licensee.

The composition of the committee specified in TS 6.2.1.1 states specifically that the permanent committee shall be comprised of the following:

1. Radiation Safety Officer
2. Reactor Facility Director
3. Chairman
4. At least 1-3 non-AFRRI members

This is a minimum or 4 members. No change to TS 6.2.1.1 is warranted.

41. Proposed TS 6.2.2, "Function and Authority," does not contain a required action with respect to reporting requirements for the RRFSS to the Level 1. ANSI/ANS 15.1-2007, Section 6.2.3, "Review function," states, "A written report or minutes of the findings and recommendations of the review group shall be submitted to Level 1 and the review and audit group members in a timely manner after the review has been completed." Propose a required action for reporting requirements for TS 6.2.2, or justify why it is not necessary to do so.

    Existing TS 6.2.3.5 b states:

    "final minutes will be submitted to level one management or review. "

    "will" will be changed to "shall"

42. Proposed TS 6.2.3.1, "Alternates," uses the terms "alternate members" and "alternates." Proposed TS 6.2.1.1, Specification b.1., defines special RRFSS members (Temporary members) as "Other knowledgeable persons to serve as alternates in section 6.2.1.1.a.2.b. above as appointed by the AFRRI Licensee."

    Provide a modification to proposed TS 6.2.3.1 and proposed TS 6.2.1.1, Specification b.1., that contain terms that are defined, define the terms used, or justify why it is not necessary to do.

    The term alternate is defined in Webster's dictionary as: "someone who is chosen to take another person's place if that person is not able to be present or to do a required job" Member and alternate are common and accepted language and do not need to be further defined in TS. No change to the TS is warranted.

43. Proposed TS 6.2.3.3, "Quorum," states, in part, that "A majority of those present shall be regular members." Proposed TS 6.2.3.4, "Voting Rules," states, in part, that "The majority is 51% or more of the regular and special members present and voting and concurrence between the Radiation Safety Officer and the reactor facility director."

    ANSI/ANS 15.1-2007 guidance states that for a quorum to be reached, not less than one-half of the voting membership shall vote, and the operating staff does not constitute a majority.

    In question 35 of the AFRRI RAI responses dated October 20, 2011, AFRRI stated:

    > Under TS 6.2.1.1, the Reactor Facility Director is the only regular member who is a member of the "operating staff." Theoretically, other members of the operating staff could be appointed as special voting members. But since under TS 6.2.3.3 a majority of those present and voting must be regular members, the operating staff can never constitute a majority. Under TS 6.2.3.3, four specific people must be present for a quorum. Depending on the number of outside regular members under TS

6.2.1.1.a(2)(b), the total regular membership could be 4-6 people. In either case, the four required people would be not less than one-half of the voting membership. In rare cases, special voting members may be appointed under TS 6.2.1.1.b. Since, as discussed earlier, a majority of those present and voting must be at least the four specific regular members, no more than three special members can be appointed for any one meeting. Here also, the four required people would be not less than one-half of the voting membership. The TS as written satisfy the ANSI/ANS-15.1 requirements.

A. Propose a clarification to proposed TS 6.2.3.3 that identifies clearly and explicitly that the guidance is met such that the operating staff does not constitute a majority, or state why it is unnecessary to provide this clarification.

6.2.3.3 will be modified to read:

A quorum of the RRFSS for review shall consist of a minimum of four members that can vote and occupy the following positions; the Chairman (or designated alternate), the Reactor Facility Director (or designated alternate), the Radiation Safety Officer (or designated alternate), and one non-AFRRI member. A majority of those present shall be regular members. The operating staff shall not constitute a majority. A member may occupy two positions.

B. Define "concurrence" and state whether this means that the votes cast by the Radiation Safety Officer and Reactor Facility Director need to agree for any item to pass the committee. Explain why a dissimilar vote between the RSO and Director should be allowed to override a majority vote from the other members or remove this limitation from the TS.

The safety committee is an advisory body for facility management. Requiring additional concurrence does not conflict with the ANSI standard. This local requirement is in addition to the ANSI guidance. No change to TS 6.2.3.4 is warranted.

44. Proposed TS 6.2.1.1, "Composition," Specification a.2.a., states that the AFRRI Licensee shall appoint the Chairman of the RRFSS. Clarify whether the Facility Director or the Radiation Safety Officer is allowed to be the Chairman, or state why it is not necessary to do so.

A committee member may hold two slots, i.e. Reactor Facility Director also acting as a chairman, but no single member may have more than one vote. See answer to 43A above.

45. The following questions pertain to the use of shall statements, or are editorial questions pertaining to grammatical or spelling:

A. Proposed TS 6.2.3.5, "Minutes," Specification b., states, "Once approved by the committee, final minutes will be submitted to level one management for review." Provide proposed TS 6.2.3.5, Specification b., in the form of a "shall" statement, or justify why it is not necessary to do so.

Provide a correction to the spelling of the word "minuites" or clarify what the term "minuites" means.

> "Will" statements will be changed to "shall". Minuites will be changed to minutes.

B. Proposed TS 6.2.2.1, "Function," states:

> The RRFSS *is* [emphasis added] directly responsible to the AFRRI Licensee. The committee shall review all radiological health and safety matters concerning the reactor and its associated equipment, the structural reactor facility, and those items listed in Section 6.2.4.

Propose a modification to proposed TS 6.2.2.1 in the form of a "shall" statement, or justify why it is not necessary to do so.

> In this context the word "is" does not denote a command or action, but is descriptive. "shall" is not appropriate in this context.

C. Proposed TS 6.7.1, "Records to be Retained for a Period of at Least Five Years," and proposed TS 6.7.3, "Records to be Retained for the Life of the Facility," are not stated with shall statements.

Revise proposed TS 6.7.1 and proposed TS 6.7.3 to include a "shall" statement or state why it is not necessary to do so.

> The title in 6.7.1 Records to be retained for a period of ... will be changed to "Records that shall be retained for a period ...

46. The following questions pertain to proposed TS 6.2.4, "Review Function."

A. Proposed TS 6.2.4, "Review Function," does not closely resemble ANSI/ANS 15.1-2007 guidance. For instance, proposed TS 6.2.4, "Review Function," Specification a., states, in part, that the RRFSS shall review:

> Safety evaluations for (1) changes to procedures, equipment, or systems having safety significance . . .

The regulations in 10 CFR 50.59 are not restricted in applicability to changes that are safety significant. Provide an explanation of how proposed TS 6.2.4 and ANSI/ANS 15.1-2007 are functionally equivalent, provide revisions to proposed TS 6.2.4 which more closely follow the guidance in ANSI/ANS 15.1-2007, or explain why it is not necessary to do so.

The proposed TS 6.2.4 are functionally equivalent to ANSI 15.1-2007. No modification to TS is warranted.

B. Proposed TS 6.2.4, "Review Function," Specification c., states that the RRFSS shall review:

> Proposed tests or experiments that are significantly different from previously approved tests or experiments, or those that might meet any of the criteria in paragraph (c)(2) of Section 50.59 of 10 CFR Part 50.

The phrase "experiments that are significantly different from previously approved tests or experiments" is used; however, no definition is provided. Provide criteria defining "experiments that are significantly different from previously approved tests or experiments," or explain why it is not necessary.

C. Explain whether AFRRI will comply with 10 CFR 50.59 in its entirety, or paragraph (c)(2) of Section 50.59 of 10 CFR Part 50.

TS 6.2.4 will be changed to read:

6.2.4. <u>REVIEW FUNCTION</u>

The RRFSS shall review:

a. Safety evaluations for (1) changes to procedures, equipment, or systems having safety significance and (2) tests or experiments conducted without NRC approval under provisions of Section 50.59 of 10 CFR.

b. Changes to procedures, equipment, or systems that change the original intent or use, are non-conservative, or those that meet any of the applicable criteria in Section 50.59 of 10 CFR;

c. Proposed tests or experiments that could affect reactivity or result in the uncontrolled release of radioactivity, or those that might meet any of the applicable criteria in Section 50.59 of 10 CFR;

d. Proposed changes in technical specifications, the Safety Analysis Report, or other license conditions;

e. Violations of applicable statutes, codes, regulations, orders, technical specifications, license requirements, or of internal procedures or instructions having safety significance;

f. Operating abnormalities having safety significance;

g. Events that have been reported to the NRC; and

h. Audit reports of the reactor facility operations.

47. ANSI/ANS 15.1-2007 guidance for TS 6.2.3, "Review Function," Item (5), states, in part, that the following items shall be reviewed: "violations of technical specifications, licenses, or charter. Violations of internal procedures or instructions having safety significance."

Proposed TS 6.2.4, "Review Function," Specification e., states, "Violations of applicable statutes, codes, regulations, orders, technical specifications, license requirements, or of internal procedures or instructions having nuclear safety significance."

The term "nuclear safety significance" is used; however, no definition is provided. Clarify whether the term "nuclear safety significance" in proposed TS 6.2.4, is the same as the term "safety significance" used in ANSI/ANS 15.1-2007, Section 6.2.3, Item (5) or explain how these are different. Propose a revision to TS 6.2.4, Specification e., that is inclusive of other types of safety, including radiation safety, or explain why it is not necessary to do so.

See 46c.

48. ANSI/ANS 15.1-2007 guidance for TS 6.2.3, "Review Function," Item (3), states, "all new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity."

Proposed TS 6.2.4, "Review Function," Specification f., does not specify the release of radiation.

Propose a revision to TS 6.2.4, Specification f., that includes the release of radiation, or explain why it is not necessary to do so.

See 46c

49. ANSI/ANS 15.1-2007 guidance for TS 6.2.3, "Review Function," Item (8), states, "audit reports [shall be reviewed]."

Proposed TS 6.2.4, "Review Function," Specification h., specifies that audit reports of the reactor facility operations shall be reviewed.

Provide a revision to proposed TS 6.2.4, Specification h., that conforms to the guidance, or provide an explanation why it is unnecessary to do so.

See 46c

50. Proposed TS 6.2.5, "Audit Function," states, in part, that "A report of the findings and recommendations resulting from the audit shall be submitted to the AFRRI Licensee,"

but does not specify the timing.

ANSI/ANS 15.1-2007, TS 6.2.4, "Audit Function," guidance states that a written report of the findings of the audit shall be submitted to Level 1 management and the review and audit group members within 3 months after the audit has been completed.

Provide a revision to proposed TS 6.2.5 that is consistent with the guidance, or justify why it is not necessary. Provide specificity as to which audits are included in reactor facility operations (i.e., whether the radiation protection program audits, TS compliance audits, etc.) are included.

This statement will be changed to read:

A report of the findings and recommendations resulting from the audit shall be submitted to the AFRRI Licensee within three months after the audit report has been received.

As stated in proposed TS 6.2.5, the following shall be audited:

a. Conformance of facility operation to the Technical Specifications and the license;

b. Performance, training, and qualifications of the reactor facility staff;

c. Results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect safety;

d. Facility emergency plan and implementing procedures;

e. Facility Physical Security Plan;

f. Any other area of facility operations considered appropriate by the RRFSS or the AFRRI Licensee; and

g. Reactor Facility ALARA Program. This program may be a section of the total AFRRI program.

51. Proposed TS 6.2.5, "Audit Function, "Specification d., does not mention the physical security plan.

NUREG 1537, Part 1, Chapter 14, Appendix 14.1, Section 6.2.4, "Audit Function," guidance states, in part, that "In addition to the emergency plan, all other required plans, such as physical security and operator requalification, should be specified for auditing."

Provide an explanation and basis for the exclusion from TS 6.2.5, or provide a revision to proposed TS 6.2.5, Specification d., that is consistent with the guidance.

See 50e. above

52. The proposed TS do not have a TS similar to the guidance in ANSI/ANS 15.1-2007 TS 6.3, "Radiation Safety;" although, parts of the guidance is followed by AFRRI in the proposed TS 6.1.2, "Responsibility."

The proposed TSs are not consistent with the guidance because there is no TS stating that an individual or group shall implement the radiation protection program using the guidelines of American National Standard, "Radiation Protection at Research Reactor Facilities," ANSI/ANS 15.11-1993 (R2004).

Either propose a TS similar to the guidance that consolidates this specification and conforms to the guidance, or modify the proposed TS so that this information is included. Alternatively, justify why it is not necessary.

Section 6.1.2. RESPONSIBILITY will be modified to read:

The AFRRI Licensee shall have license responsibility for the reactor facility. The Reactor Facility Director (RFD) shall be responsible for administration and operation of the reactor facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The Reactor Facility Director may designate an individual who meets the requirements of Technical Specifications 6.1.3.1.a to discharge these responsibilities during an extended absence. During brief absences (periods less than 4 hours) of the Reactor Facility Director and his designee, the Reactor Operations Supervisor shall discharge these responsibilities The radiation Safety Officer shall be responsible for implementing the radiation safety program for the AFRRI TRIGA reactor. The requirements of the radiation safety program are established in 10CFR20. The program shall comply with the requirements in 10CFR20. Additional guidelines from ANSI/ANS-15.11-1993;R2004 "Radiation Protection at Research Reactor Facilities" should be considered.

53. Proposed TS 6.3, "Procedures," does not propose procedures to address the use, receipt, and transfer of byproduct material or surveillance procedures for shipping radioactive materials.

NUREG-1537, Part 1 guidance includes these procedures, as does ANSI/ANS 15.1-2007, Section 6.4, "Procedures," Item (8).

Propose a TS that includes these procedures or justify why it is not necessary.

All byproduct materials are transferred to the AFRRI O2 license. Shipping, receipt and transfer of byproduct materials is handled under the O2 license.

No changes to TS are warranted.

54. Proposed TS 6.5.1, "Actions to be Taken in Case of Safety Limit Violation," is only partially consistent with ANSI/ANS 15.1-2007 with respect to reporting criteria for safety limit violations. Proposed TS 6.5.1 does not include the phrases, "when known, the cause, and contributing factors;" or "structures, or systems, the health and safety of personnel and the public."

ANSI/ANS 15.1-2007 guidance uses these reporting criteria.

Propose a revision to proposed TS 6.5.1 that is consistent with the guidance and contains these reporting criteria or provide additional information to explain how proposed TS 6.5.1 is consistent with the guidance.

6.5.1 will be modified to read:

6.5.1. <u>ACTIONS TO BE TAKEN IN CASE OF SAFETY LIMIT VIOLATION</u>

a. The reactor shall be shut down immediately, and reactor operation shall not be resumed without authorization by the USNRC.

b. The safety limit violation shall be reported to the USNRC, the AFRRI Licensee, and the RRFSS not later than the next working day.

c. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the RRFSS, and shall describe (1) applicable circumstances preceding the violation, when known, the cause and contributing factors (2) effects of the violation on facility components, structures, or systems, the health and safety of personnel and the public and (3) corrective action taken to prevent or reduce the probability of recurrence.

d. The Safety Limit Violation Report shall be submitted to the USNRC, the AFRRI Licensee, and the RRFSS within 14 days of the violation.

55. Proposed TS 6.5.2, "Reportable Occurrences," Specification e., discusses requirements for reporting observed inadequacies that could have caused unsafe conditions. However, no mention is made of requirements for reporting observed inadequacies that are *currently* causing unsafe conditions, as ANSI/ANS 15.1-2007 guidance states.

Provide a revision to proposed TS 6.5.2 that requires reporting observed inadequacies that are *currently* causing unsafe conditions, or justify why it is unnecessary.

6.5.2.e will be modified to read:

e. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of a condition that could result in operation of the reactor in a manner less safe than conditions covered in the Safety Analysis Report.

56. The following questions pertain to proposed TS 6.6, "Operating Reports":

A. Proposed TS 6.6, "Operating Reports," Specification b.4., states that each annual operating report shall include: "Discussion of the major safety-related corrective maintenance performed during the period, including the effects (if any) on the safe operation of the reactor, and the reasons for the corrective maintenance required;" however, the tabulation of preventative maintenance operations having safety significance is not required by the proposed TS.

Revise proposed TS 6.6, Specification b.4., to be consistent with ANSI/ANS 15.1-2007 guidance which includes preventative maintenance operations having safety significance or justify why it is unnecessary.

TS 6.6 specification b.4 will be modified to read

Discussion of the major safety significant corrective and/or preventive maintenance performed during the period, including the effects (if any) on the safe operation of the reactor, and the reasons for the corrective maintenance required;

B. Proposed TS 6.6, Specification b.4. contains the term "safety-related corrective maintenance," but does not include a definition. ANSI/ANS 15.1-2007, Section 6.7.1, "Operating Reports," Item (3) uses the term "safety significance." The term "safety-related" has a specific definition provided in 10 CFR Part 50, Appendix B. However, 10 CFR Part 50, Appendix B is not applicable to non-power reactors. Revise proposed TS 6.6, Specification b.4., using terms applicable to your facility [e.g., safety significant.] Explain how proposed TS 6.6, Specification b.4., is consistent with the guidance, or propose a revision that is consistent with the guidance.

TS 6.6 b.4. will be changed to read:

4. Discussion of the major safety significant corrective and/or preventive maintenance performed during the period, including the effects (if any) on

the safe operation of the reactor, and the reasons for the corrective maintenance required;

57. Proposed TS 6.6, Specification c., "Other Reports," states that "A report shall be submitted within 30 days ..." but does not specify to whom the report should be submitted. To be consistent with ANSI/ANS 15.1-2007 guidance, specify the required recipient of the report, or justify why it is unnecessary.

TS 6.6, Specification c. will be modified to read:

b. Other reports: A report shall be submitted to the USNRC within 30 days describing:

58. On February 7, 2011, AFRRI responded to the NRC staff's RAI, dated July 19, 2010. Question 9 asked AFRRI to demonstrate that external event consequences show compliance with the regulations in 10 CFR Part 20. AFRRI's response stated, "Therefore, seismic events in the D.C. area are not a viable threat to the integrity of the AFRRI reactor facility." In consideration of the 5.6 magnitude earthquake that struck 14km SSE of Louisa, VA on August 23, 2011, and was detectable in Bethesda, MD, state whether your response to the July 19, 2010, RAI question 9, remains valid (i.e., remains bounded by the seismic analysis of record for the AFRRI reactor facility). If not, provide the necessary revision.

The response remains valid.

59. NUREG-1537, Part 1, Chapter 13, Section 13.2, "Accident Analysis and Determination of Consequences," states that when evaluating accident consequences, licensees should account for consequences that could occur to reactor staff or members of the public until the accident has terminated, or these individuals have been evacuated or moved.

In its RAI dated July 19, 2010, the NRC staff requested that AFRRI provide accumulated doses to the reactor building occupants and to the maximally exposed member of the public, considering evacuation procedures and potential residence time for staff, and asked AFRRI to discuss compliance of these doses with the regulations in 10 CFR Part 20.

In its response, dated January 17, 2012, AFRRI analyzed doses to individuals ("Receptor A," "Receptor B," and "Receptor C") from external radiation originating from airborne radioactive material inside the reactor room. AFRRI's response stated, in part, that:

> ...Receptor C is located 100 ft. from any reactor wall, with an additional concrete block wall between Receptor C and the reactor wall...

...Receptor C represents the closest location of an emergency evacuation assemblage point. For the purposes of this calculation, it was assumed that a member of the public could stay at this assemblage point for 2 hours following the accident. In reality, personnel would be evacuated to a more distant location in this type of accident...

A. Clarify whether Receptor C represents a member of the AFRRI staff evacuated from the AFRRI facility, or a member of the public.

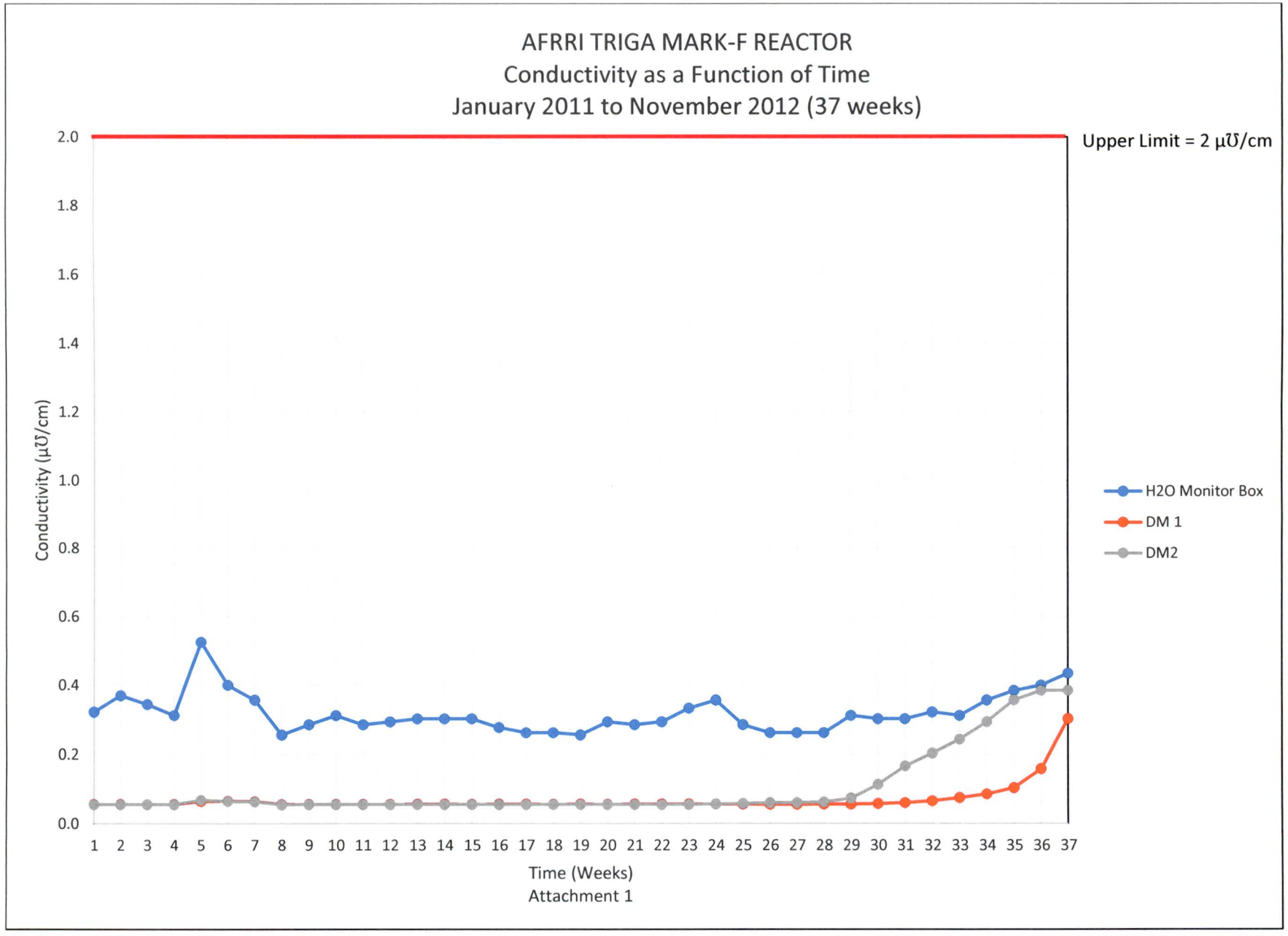
Receptor A is an AFRRI reactor staff member and a radiation worker.

Receptor B is an AFRRI staff member but not a reactor staff member and is not necessarily a radiation worker

Receptor C is an AFRRI staff member. For the purposes of this calculation, an AFRRI staff member would be considered to be a member of the public, but would be under direct supervision of the licensee in an area under direct control of the licensee..

B. The emergency evacuation assemblage points for individuals evacuated from the AFRRI facility are not discussed in the AFRRI Emergency Plan. Provide a basis for the assumption that the closest emergency evacuation assemblage point would be 100 feet from any reactor wall, with an additional concrete block wall between Receptor C and the reactor wall. Additionally, provide a basis for the assumption that individuals would remain at this emergency evacuation assemblage point for two hours following the accident.

The AFRRI Emergency Plan does identify the emergency muster locations for AFRRI personnel; the AFRRI Reactor Facility Emergency Plan covers emergencies for the reactor facility only. The two muster locations are at the extremes of the AFRRI site as far from the reactor building as possible, over 100 feet from any reactor wall with several intervening buildings and multiple concrete walls and structures.
The analysis states that a Receptor C could stay at that location for 2 hrs. and receive a dose of 4mR, not that they must stay at that location for 2hrs.



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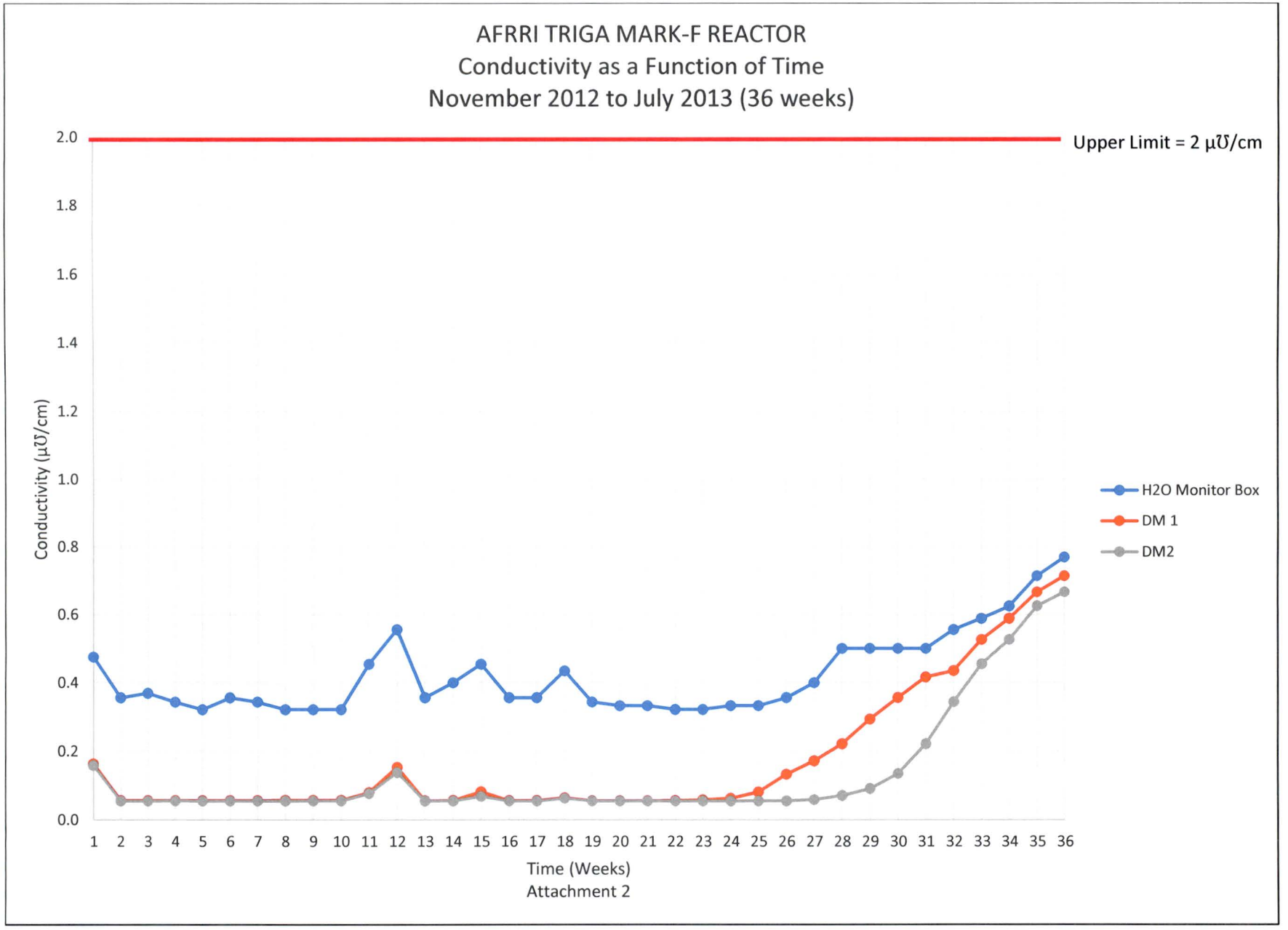

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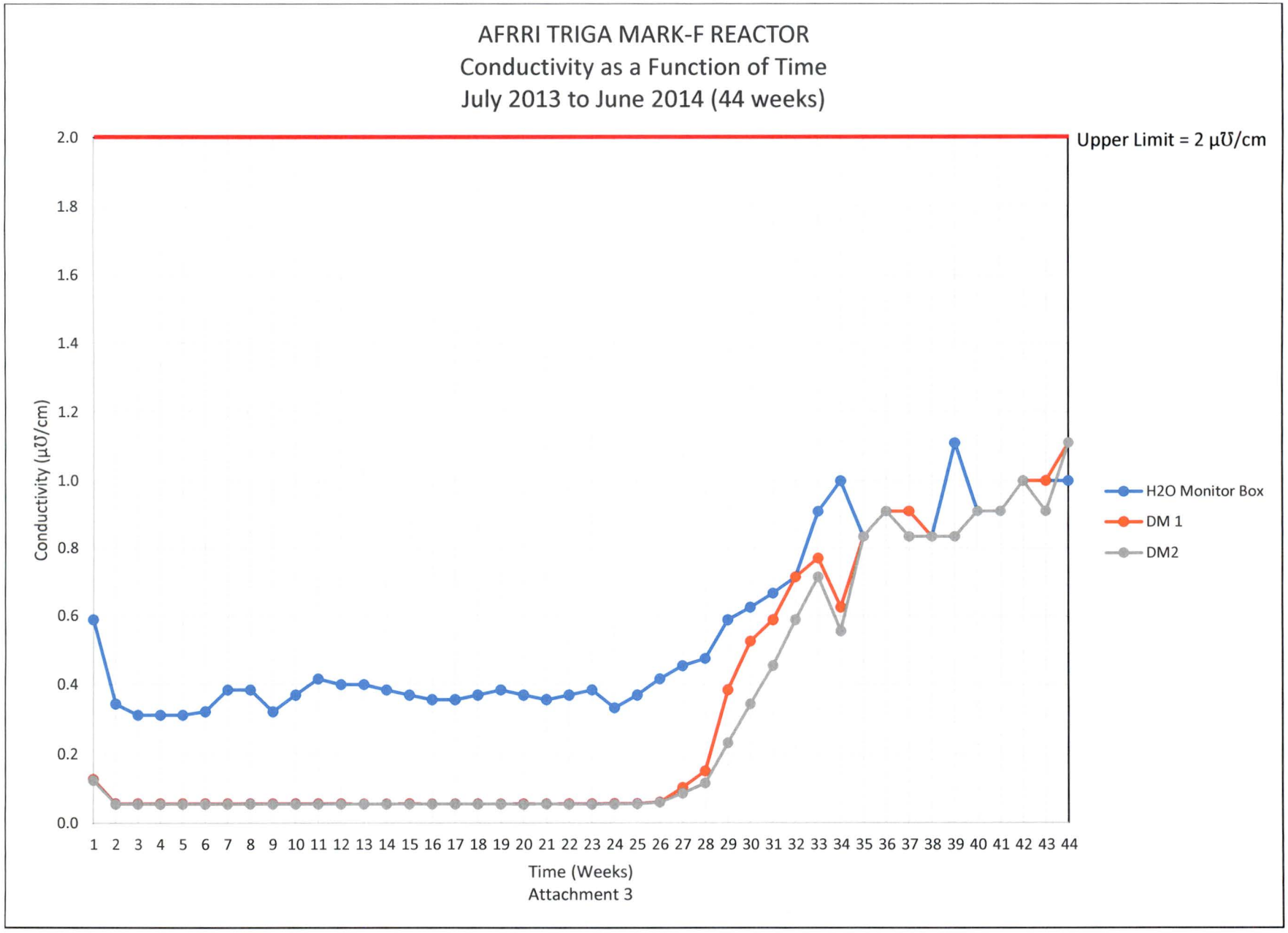

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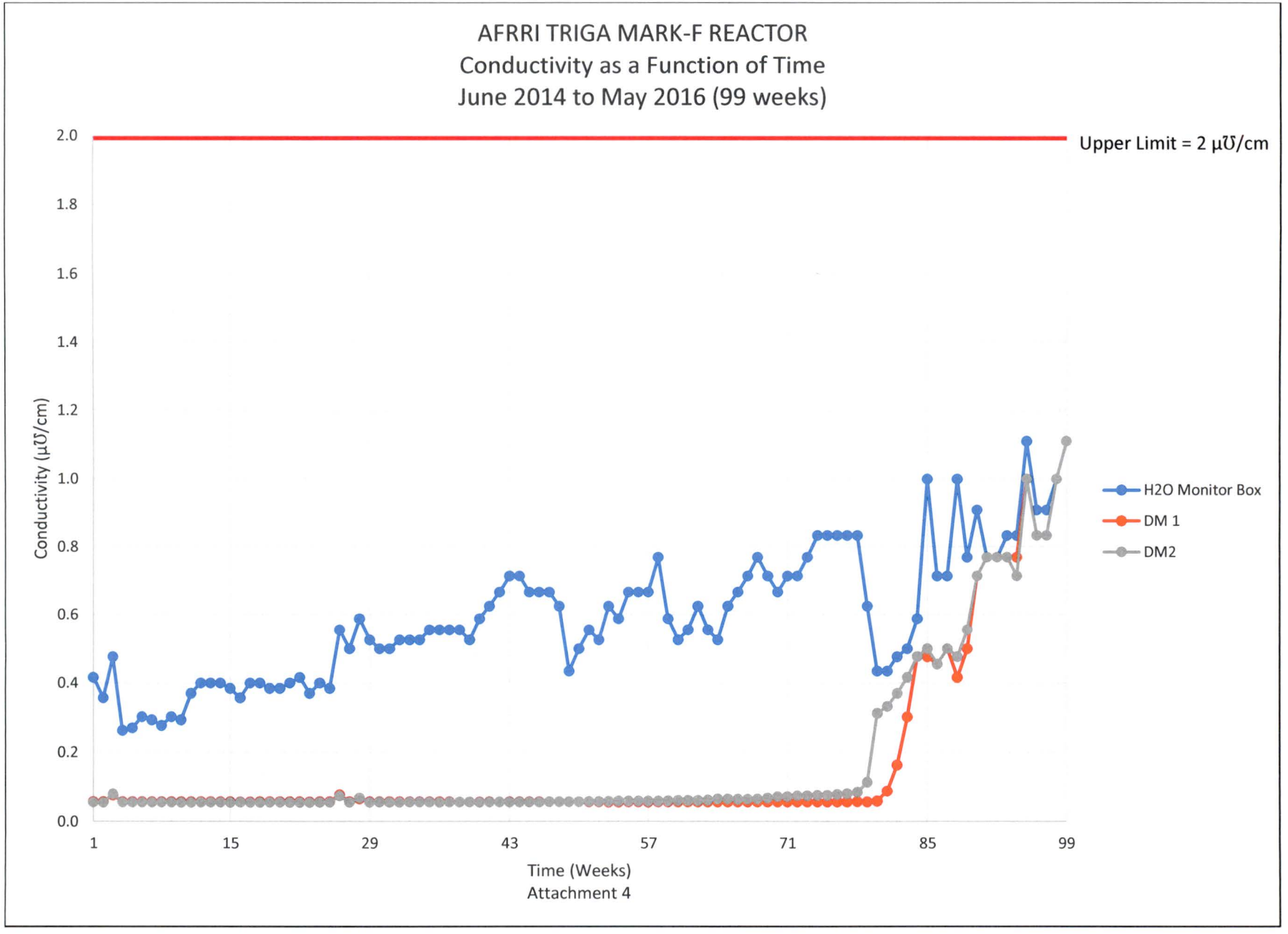

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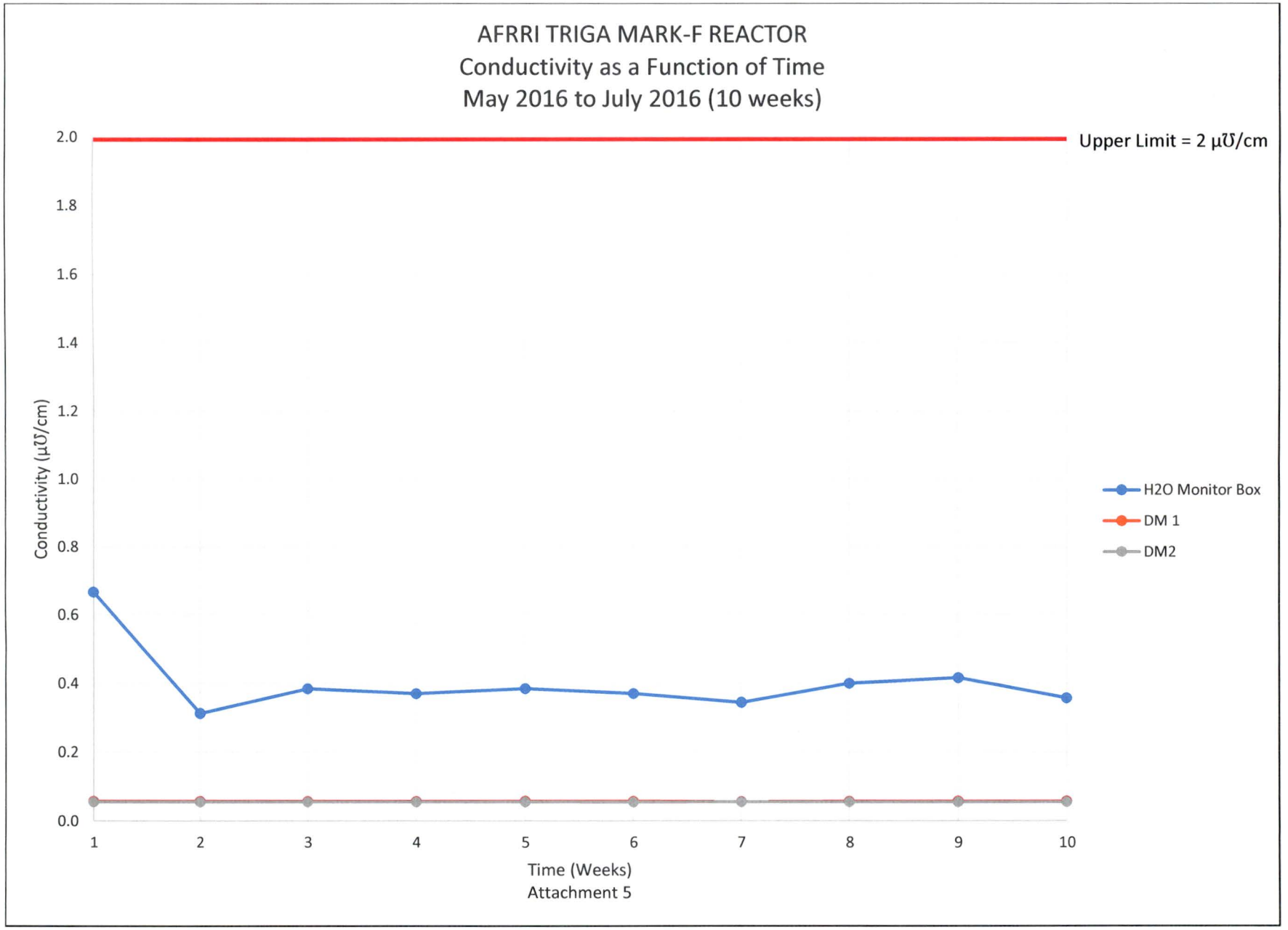

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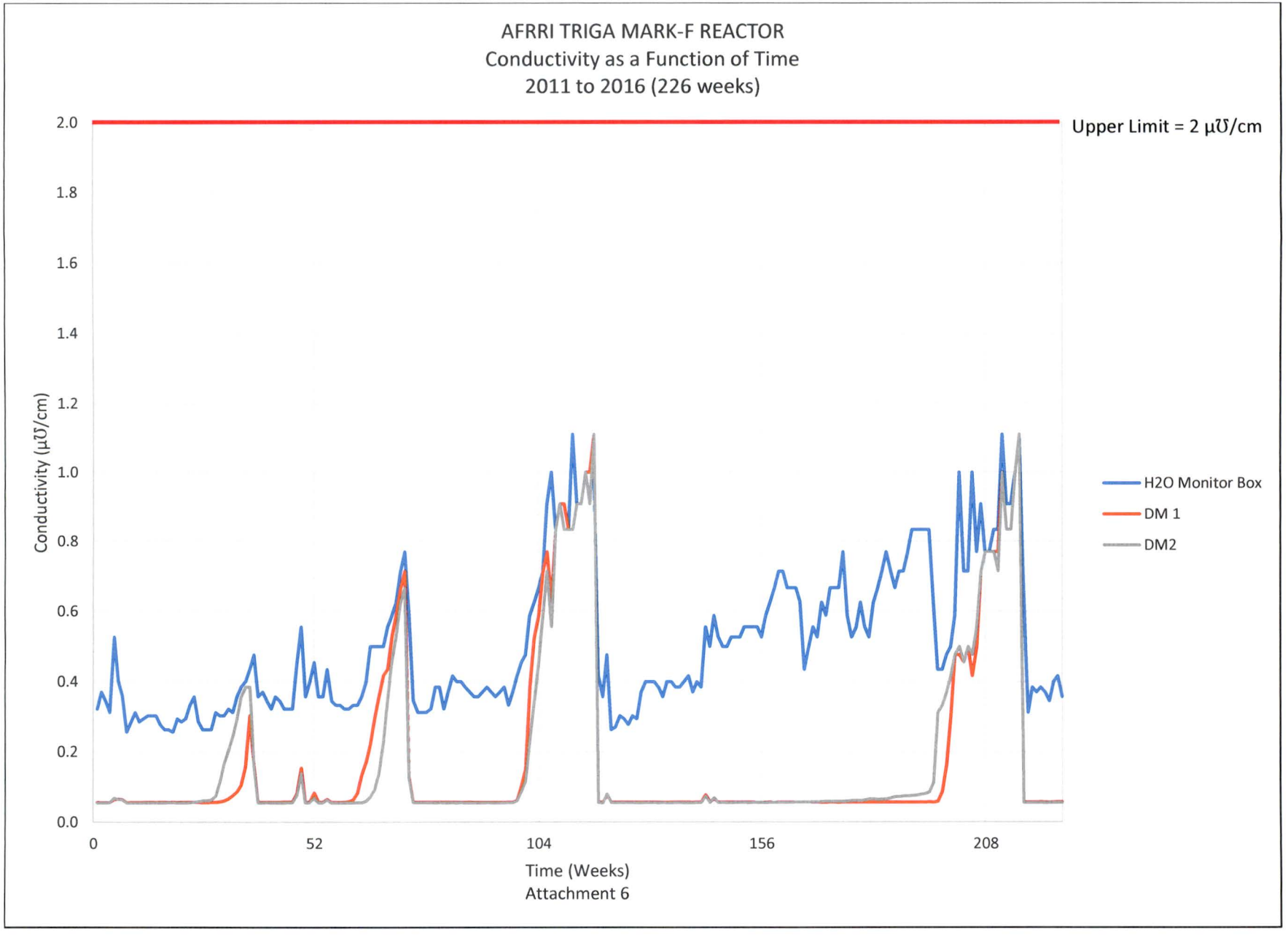

Attachment 6